NON-PUBLIC?: N
ACCESSION #: 8805030140
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 of 7

DOCKET NUMBER: 05000410

TITLE: Reactor Scram Caused by Inadequate Plant Impact Assessment Before Performing Loop Calibration on Feedwater Flow Transmitters
EVENT DATE: 03/21/88 LER #: 88-017-00 REPORT DATE: 04/20/88

OPERATING MODE: 1 POWER LEVEL: 97.5

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Robert E. Jenkins, Assistant Supervisor Technical Support
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SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: On March 21, 1988 at 1027 hours with the reactor mode switch in run (Operational Condition 1) and at a power level of approximately 97.5% rated thermal capacity, Nine Mile Point Unit 2 experienced an automatic reactor scram as a result of the main turbine trip. The turbine trip occurred when a feedwater flow transmitter was valved out of service creating increased feed flow until Level 8 was reached. (Normal turbine trip on high reactor water level.)

The root cause of the event is that current work control procedures do not assure proper assessment of plant impact.

Immediate corrective action was to restore reactor water level to normal. Further corrective actions include revision of repair and trouble shooting procedures, plant impact policy issuance, lessons learned transmittals, and system design review.

(End of Abstract)

TEXT: PAGE: 2 of 7

I. DESCRIPTION OF EVENT

On March 21, 1988 at 1027 hours with the reactor mode switch in run (Operational Condition 1) and at a power level of approximately 97.5% rated thermal capacity, Nine Mile Point Unit 2 (NMP2) experienced an automatic reactor scram as a result of a main turbine trip. Approximately 20 seconds prior to this, two Niagara Mohawk Instrument and Control Technicians (I&C Technicians) were valving feedwater flow transmitter 2FWS-FT1B out of service to perform a loop calibration on TL2FWS-086 (see Diagram #1). A work request had been processed per the administrative procedure for repair (AP-5.2).

The feedwater control system (FWS) responded as designed. It sensed a steam flow/feed flow mismatch and increased feed flow to the reactor by automatically opening the in service feedwater control valves further (LV10A & C - see Diagram #2). High reactor vessel water Level 8 (202.3") was reached at approximately 1027 hours and signals from the reactor vessel water level instrumentation then resulted in a trip of the main turbine and feedwater pumps. The main turbine trip initiated fast closure of the turbine control valves which subsequently initiated reactor recirculation pump high-to-low transfer sequence and reactor scram.

Following the reactor scram water level dropped to 110 inches. Niagara Mohawk control room operator response was to assess the situation, reset the feedwater pump high level trips, and restart feedwater pump 2FWS-P1A. Reactor vessel water level was restored to normal, and plant shutdown was achieved per Operating Procedure N2-OP-101C. Level was restored 12 minutes after the scram.

All systems functioned per design. There were no inoperable structures, components, or systems that contributed to this event.

II. CAUSE OF EVENT

The immediate cause for the scram was valving out of service a feedwater flow transmitter by two I&C Technicians. The root cause is that the administrative procedure for repair (AP-5.2) did not provide adequate guidance for work control or require plant impact statements to accompany work requests (WRs). When the I&C Technicians valved out the transmitter, the resulting plant response was unanticipated by Operations personnel. AP-5.2 is effectively silent on identifying the responsible group for determining plant impact. The following is a list of work control process steps which could have prevented the event. However, AP-5.2 does not identify any of the noted groups as responsible for plant impact determination and job control.

1. The originator of the WR was qualified to assess plant impact and stipulate under what conditions the work could be performed. The WR originator did not communicate the potential plant impact of performing the requested work.

2. Although technically correct, the drawings used to assess plant impact did not describe the complete plant effects without significant research. More attention to detail on the part of the I&C Technicians and the SSS reviewing this drawing could have avoided an incorrect assessment of plant impact.

3. Had I&C Technicians with a high degree of familiarity with the feedwater system been selected to perform this job, the probability of a correct plant impact assessment would have been enhanced. However, I&C Technicians with limited feedwater system experience were assigned to do the work. They did not recognize the effect on the system of performing the work.

4. The WR was written while the plant was shutdown and was prio
itized as
"urgent" (to be worked within one day). Had the work been completed the first day as assigned; the event would not have occurred.

5. At the daily work control meeting three days after the WR was approved by the ASSS, during plant startup, the WR was reprioritized to be performed at a later date. I&C supervision did not ensure that the WR was retrieved from the work crew, and thus did not prevent the work from being initiated.

6. The SSS reviewing the WR could have requested additional review when realizing that the feedwater system was involved and the work item was not on the work instructions for the day.

7. I&C supervision should have recognized the potential plant impact and placed restrictions on performing the work.

III. ANALYSIS OF EVENT

There were no adverse safety consequences as a result of this event. Section 15.1.2 of the NMP2 Final Safety Analysis Report (FSAR) is an accident analysis for "Feedwater Controller Failure - Maximum Demand". The event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow. The expected plant response to this is high water level turbine trip and feedwater pump trip. The scram occurs simultaneously and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. The plant responded as analyzed. Twelve minutes elapsed from the scram to water level restoration.

IV. CORRECTIVE ACTIONS

1. Immediate corrective action was for the NMPC control room operators to regain normal reactor vessel water level and proceed to shutdown per Operating Procedure N2-OP-101C.

2. The Technical Superintendent will revise the administrative procedure for repairs AP-5.2 and the administrative procedure for temporary modifications AP-3.3.2 to clearly address plant impact requirements by September 1, 1988. Input from appropriate user departments including Operations and I&C will be solicited. Specific attention will be given to the work control processes addressed in the Cause of Event section of this text.

3. In the interim (until the procedure is revised) the following corrective actions have been taken:

a. The Electrical, Computer, and I&C Departments have implemented a new policy of attaching a plant impact review form to work documents without plant impact already incorporated. This form specifically addresses plant impact, reference documents, permissible reactor operational mode, and review by supervision or the chief technician.

b. A Lessons Learned Transmittal has been written to Operations, I&C, and Electrical and Mechanical Maintenance stating the work control processes that could have prevented the scram and advising responsible individuals to respond accordingly.

c. By management directives, administrative controls for work control and authorization of work to be performed will be issued to all departments.

4. Engineering is reviewing current system design to determine future action if necessary. By design, a loss of feedwater flow signal (one transmitter) causes a bias of 25 inches in sensed narrow range level. As Level 8 trip is 202.3 inches and normal level is 183 inches, a bias of 25 inches will cause Level 8 to be reached prior to water level reaching equilibrium. Contrary to this design, it is desirable to maintain the reactor in operation in the event of a single feed flow signal

failure. Problem Report 07830 has been written recommending a design change that would minimize the impact of feed flow or steam flow signal failures with feedwater control in three element auto.

V. ADDITIONAL INFORMATION

A. Identification of Components Referred to in this LER

IEEE 803 IEEE 805
Component EIIS Funct System ID

Reactor RCT N/A
Turbine TRB N/A
Flow Transmitter FT SJ
Feedwater Control System (FWS) --- SJ
Flow Control Valves FLV SJ
Feedwater Pumps P SJ
Level Instrumentation LIT SJ
Reactor Recirculation Pump P AD
Turbine Control Valves FCV TA

B. There are no previous events where a scram resulted from an erroneous feedwater flow signal. There are previous events whose cause was at least partially related to plant impact definition for non-procedure related work control. These are LER# 87-17, 87-26, 87-64, 88-06. The corrective actions for these LER's addressed immediate and problem specific causes and did not focus on the administrative procedure for repairs (AP-5.2).

C. Failed Components - None

DIAGRAM #1

FIGURE OMITTED - NOT KEYABLE (DRAWING)

DIAGRAM #2

FIGURE OMITTED - NOT KEYABLE (DRAWING)

ATTACHMENT # 1 TO ANO # 8805030140 PAGE: 1 of 1

NIAGARA NMP32828

MOHAWK

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April 20, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 88-17

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

LER 88-17 Is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

A 10CFR50.72 report was made at 1117 hours on March 21, 1988.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,
/s/ Thomas J Perkins
Thomas J. Perkins
Vice President - Nuclear

TJP/DRG/mjd

Attachments

cc: Regional Administrator, Region 1
Sr. Resident Inspector, W. A. Cook

PO24730304

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